U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-244/93-07

Docket No. 50-244

License No. DPR-18

Licensee: Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Facility Name: Ginna Nuclear Power Plant

Inspection At: Ontario, New York

Inspection Conducted: April 12 - 16, 1993

Inspectors:	J. Noggle J. Noggle, Sr. Radiation Specialist	5/17/93 date
	J. Noggle for J. Nick, Radiation Specialist	5/17/93 date
Approved by:	James H. Joyner W. Pasciak, Chief, Facilities Radiation Protection Section, DRSS	5/17/93 date

Areas Inspected: The inspection was an announced inspection of the radiological controls program implementation during outage conditions.

Results: During the inspection, the containment was inaccessible due to the performance of an integrated leak rate test and limited work was performed in other radiological work areas. Therefore, the inspection did not include a representative review of HP coverage of work in progress nor was there opportunity to ascertain the adequacy of the air sampling program. Instead, records were reviewed, and discussions with licensee personnel were held to provide a measure of outage HP performance within the limited context of the inspection. Radiological postings and surveys were strong. Special Work Permits, which specify the radiological controls for workers, were not always detailed but were generally well written.

There was evidence of continued efforts made in the temporary shielding program with the need for further improvements noted. The air sample tracking system has been improved. Radiological controls for steam generator work were well applied, with resulting low exposures.

DETAILS

1.0 Personnel Contacted

1.1 Licensee Personnel

J. Bement, Health Physics Operations Lead Technician
* J. Bettle, Preventive Maintenance Engineer
* J. Cook, Reactor Engineer
* J. Fischer, Maintenance Planning and Scheduling Manager
* W. Goodman, Health Physics Operations Foreman
* T. Harding, Modification Support Coordinator
* A. Harhay, Health Physics and Chemistry Manager
* A. Herman, Health Physicist
J. Johnston, HP Technician Instructor
* J. Knorr, Supervisor, HP/Chemistry Training
K. Lang, Health Physicist
* N. Leoni, Quality Improvement Specialist
* R. Marchionda, Superintendent, Support Services
* F. Mis, Health Physicist
* R. Ploof, Technical Engineer
* B. Quinn, Corporate Health Physicist
* W. Rapin, Modification Support Engineer
* J. St. Martin, Corrective Action Coordinator
* T. Schuler, Operations Manager
M. Smith, Steam Generator Project Engineer
* W. Thomson, Health Physicist
R. Watts, Director, Corporate Radiation Protection
* J. Wayland, Reactor Engineer
L. Weaver, Steam Generator ALARA** Technician
* J. Widay, Plant Manager
J. Wright, ALARA HP Technician
* G. Wrobel, Manager, Nuclear Safety and Licensing

1.2 NRC Personnel

E. Knutson, Resident Inspector
* T. Moslak, Senior Resident Inspector

* Denotes attendance at the exit meeting on April 16, 1993.
** ALARA = As Low As is Reasonably Achievable

2.0 Organization

The permanent Health Physics (HP) organization consists of approximately 45 personnel. There are currently four staff Health Physicists who are responsible for individual program areas and a Quality Improvement Specialist. These individuals report directly to the HP and Chemistry Manager.

The outage HP organization was expanded with the addition of a temporary workforce of 56 contractor HP technicians, consisting of 39 senior technicians and 17 junior technicians. The permanent staff Health Physicists were assigned different shifts and provided continuous HP supervision during the outage. There were three satellite HP checkpoints as well as the HP office "desk" where plant areas of responsibility were subdivided. The HP "desk" provided the focus and job coverage responsibility for work in the auxiliary building and on the refueling floor inside containment. There were 3 separate HP checkpoints outside the crane wall located on the containment basement (46 foot) level responsible for "A" steam generator, "B" steam generator and "other" containment work below the refueling floor elevation. Permanent station senior HP technicians provided the lead HP technician functions for these checkpoints. This level of staffing appeared to be commensurate with the outage work demands. Appropriate levels of first-line and management supervision were evident.

3.0 Training and Qualifications

The licensee's training and qualification program for contractor HP technicians was reviewed through discussions with personnel and through the review of qualification records. Criteria used for this review included the American National Standard ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," NUREG-1220,"Training Review Criteria and Procedures," and the licensee's training program procedures.

Through a sampling of contractor HP technician resumes, the licensee was found to be in compliance with the experience requirements of ANSI N18.1-1971. The inspector also examined the training program for the temporary contractor HP technicians. The licensee has adopted an abbreviated training program, utilizing an incoming HP exam to determine basic HP knowledge of the HP technician applicant as a substitute for providing this training. The licensee is a member of the Middle Atlantic Nuclear Training Group (MANTG), which sponsors a bank of exams that have been reviewed and approved for use by participating licensees. The administration of the MANTG exam was used to qualify HP technicians for generic HP technician responsibilities. A site specific training course was provided, consisting of approximately 16 subjects presented in 16 - 32 hours of classroom training, resulting in an examination to fully qualify HP technicians on Ginna specific procedures. The inspector reviewed the results of both the MANTG generic HP exam and the Ginna site specific HP exam for selected HP contractor technicians and noted that several senior HP technicians had received scores of less than 80% (the station's and MANTG's pass/fail criteria) after two attempts to pass the MANTG exam and were qualified to perform senior HP technician duties by waiving of this requirement by a staff Health Physicist. Upon further review, it was determined that the licensee had not officially adopted acceptance criteria relative to the MANTG HP technician exam, and had not utilized a training review board to waive individuals from successfully passing this exam, as required by station procedure. The licensee agreed to revise the appropriate procedure incorporating a specific MANTG exam acceptance criteria.

4.0 Radiological Incident Reports

The inspector reviewed five Radiological Incident Reports (RIRs) recorded by the licensee for the current outage. Except for one RIR, all the RIRs characterized events of minor safety significance. The one incident involved the response to a small fire in containment on March 19, 1993, when the HEPA exhaust hose from the "A" steam generator tent was displaced from its normal alignment to the intake plenum of the containment air filtration system, which contained charcoal adsorbers that normally remove radioactive iodine. Iodine airborne activity increased inside containment until the source of the problem was identified and the correct ventilation alignment was established. Appropriate MPC-hour assignments were made to all personnel inside containment during the subject time period, resulting in a maximum internal exposure assessment of 7 MPC-hours (versus 520 MPC-hours/ quarter limit). Other RIRs included documentation of repetitive poor worker practice, poor reactor cavity water clarity, and documentation of a good HP response to a radiography evolution. No programmatic problems were identified from the review of the documented radiological incidents and licensee processing of the reports was thorough.

5.0 Radiological Conditions

The inspector reviewed the radiological work conditions present at the time of the inspection to allow an assessment of applied radiological controls by the licensee to protect the health and safety of station workers. The inspector reviewed this area through discussions with personnel, by review of current surveys provided by the licensee, and by verification of dose rates in many of the outage work areas.

Radiological hazards in the workplace were evaluated based on the following:

Location	Radiation mR/hr	Contamination dpm100 cm^2
Containment refuel level	1-4	1K-2K (K=1000)
Reactor cavity, post-decon	20-30	20K-300K
Containment intermediate level	1-5	<1K,50K seal table
Containment basement level	1-10, 30-80 RCS	2K, 50K occasional
Steam Generator primary platforms	20-50	10K-100 mrad/hr/100cm^2
Steam Generator primary entries	6-9 R/hr	
Steam Generator secondary platforms	5-60	<1K-3K
Steam Generator secondary entries	8-15 R/hr	
Reactor Coolant Pump platforms	5-45	1K-5K
Pressurizer, top	15-50	20K
Pressurizer, intermediate	10-50	2K-180K

The licensee produced an appropriate number of surveys and they were generally of good quality. Air survey results indicated low levels of radioactive contamination were generally maintained. It should be noted that the air sample tracking system has been automated, which has eliminated the calculation errors that were present in the past. Internal exposure assessment from air surveys still requires manual correlation of personnel occupancy times with the air surveys and requisite calculations. The real-time access control/dosimetry system under consideration by the licensee would further improve this area.

6.0 Radiological Controls

The inspector reviewed the implementation and adequacy of radiological controls with respect to radiological conditions present during outage conditions. The evaluation of the licensee's performance was based on discussions with cognizant personnel, review of inspector selected documents, and limited observations of on-going work activities during tours of Ginna Station. The inspector made entries into the reactor coolant system loop areas to verify radiological postings and adequacy of licensee controls. In general, the containment and auxiliary building areas were well posted with sufficient radiological information to warn the worker of the existing radiological hazards. No significant deficiencies in postings or housekeeping were noted.

6.1 Special Work Permits

The following SWP's were used during the outage and were reviewed by the inspector:

SWP	Job	Comment
174	Radiography in Aux Bldg Annex	OK
204	Repack Charging Pump B	OK
226	Remove/Replace Insulation	No req. for post-remove survey No precautions for handling asbestos
232	1B RCP Major Pump Seal Inspect	OK
247	Perform NDE in Pressurizer area	No precautions for P/T
306	ALARA Field Activities	OK
445	Reactor Head Reassembly	No relocation of dosimetry specified
353	Install plugs in B Steam Generator	OK
362	Tube sleeving in B Steam Generator	OK
448	Decon Reactor Cavity	OK
266	Reactor Head Lift and Cavity Flood	OK

The licensee had improved the format of the SWPs since the last inspection. The inspector found the SWPs to be generally well written, and to prescribe good general radiological controls and ALARA instructions to ensure protection of the worker.

6.2 Steam Generator Job Coverage

Steam generator associated work (not including insulation replacement) resulted in the largest exposure for the outage, representing approximately 45% of the total exposure. The outage work had been completed at the time of the inspection; however radiological controls were reviewed through discussions with the licensee and through the review of records.

Approximately 51 person-rem was expended in the performance of remotely performed eddy current tube inspection work, and during the installation of approximately 300 tube sleeves and several tube plugs into the two Ginna steam generators. The highest exposure to an individual was 1.5 rem to the whole body and 3.9 rem to the extremities (representing 50% and 20% of regulatory limits).

The licensee operates a steam generator building with two steam generator mockups available for equipment checkout and for training of work crews. This building also houses offices to support the project engineer and staff, and includes an ALARA office.

The radiological controls setup for primary-side steam generator work was adequate. A containment tent was built on each generator platform to include the access stairway. A 2000 CFM High Efficiency Particulate Air (HEPA) filter provided air flow through the tent and air suction on the steam generator. During manned steam generator work, a laborer was normally stationed inside the shielded portion of the steam generator tent and one HP technician was stationed outside the biological shield wall at a video and audio communications control station. Two video cameras were used on each platform with one audio party line for each steam generator, which connected the steam generator worker, HP technician, steam generator building, and vendor trailer with one another. During normal robotic steam generator maintenance activities, there was no one stationed inside the biological shield wall. Manned steam generator entries were only required for the removal and replacement of nozzle dams, with platform attendance only required for installing and removing manways and inserts, staging of the robotic equipment and for changing eddy current probes or other tooling from the robot arm. The inspector was satisfied that steam generator maintenance associated exposure was minimized.

In summary, the licensee has an excellent steam generator mockup facility and very good audio and video communications between the steam generator worker, HP technician, and the steam generator building. The Special Work Permits were well written and appropriate radiological controls were applied. The licensee had staged adequate containments for contamination control and had minimized steam generator entries and platform attendance, which reduced exposures.

7.0 Personnel Contamination Reports

The station set a goal for limiting personnel contaminations to below 100 for the 1993 refueling outage. As of April 13, 1993, there were 72 personnel contamination incidents recorded which resulted in 5 skin dose assignments. The inspector noted that the number of personnel contaminations was down by 50% compared to last year's outage performance. Licensee actions that were attributed to affect these results included:

- Successful reactor cavity decontamination via hydrolazer;
- Reactor cavity workers were assisted by HP in removing protective clothing;
- Outage planning computer was used to sort work by location to help define daily decon needs;
- The use of modesty garments worn underneath the regular protective clothing was provided for decon, HP, and laundry workers as a pilot program and appeared to be effective in reducing the incidents of personnel contamination from these work groups; and
- The station switched from using an in-house drycleaning protective clothing cleaning service to a vendor supplied water wash protective clothing cleaning service. According to the licensee, this resulted in slightly cleaner protective clothing.

8.0 ALARA

The inspector reviewed controls for maintaining radiation exposures as low as is reasonably achievable (ALARA) through direct observation of the work site, review of records, and through interviews relative to criteria contained in the following:

- Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable
- Regulatory Guide 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable

8.1 ALARA Status

The licensee set an annual ALARA goal of 180 person-rem for the year of 1993. As of April 14, 1993, the licensee had recorded 4 person-rem of non-outage exposure for the year and 142 person-rem of outage exposure with less than 1 week of outage demobilization work remaining. Licensee expectations were good that they will be able to meet the annual goal. The maximum internal exposure recorded for any calendar quarter in 1992 was 9 MPC-hours, versus a regulatory limit of 520 MPC-hours/quarter.

8.2 Shielding Program

The inspector reviewed the licensee's temporary shielding program. The inspector observed various temporary shield designs used during the outage. Most of them consisted of a single layer of approximately ¼ inch lead shielding. High background levels continued to exist around many of these shielded sources. The licensee took appropriate posting precautions in some of these areas to warn workers of these high background levels, supplemented by the use of locally indicating area radiation monitors. However the shielding observed by the inspector did not appear to be sufficient for all cases. The inspector reviewed procedure A-1.7, Rev. 1, entitled, "Control of Temporary Lead Shielding" and determined it to be an effective vehicle for effecting the design, engineering approval, installation, control, and removal of temporary lead shielding, but found that it did not address shield design optimization. The licensee has added an onsite Technical Engineering group as a resource to facilitate review and approval of shielding requests as well as minor plant modifications. This onsite engineering assistance helped expedite the process. The inspector reviewed the ALARA shielding packages and assessed the development of the ALARA shielding program as below.

- RHR Suction Line. This shielding package consisted of one lead blanket thickness covering the bottom half of the pipe, covering 109 linear feet of pipe with approximately 60 lead blankets. The shielding package approval allowed 16-27 lb/ft loading without requiring a reanalysis of pipe loading. Preshielding dose rates were 10 - 50 mR/hr. Shielding effectiveness was recorded based on pipe contact readings rather than general area field dose rates. The shielding package indicated a pre-shielding maximum dose rate of 250 mR/hr down to a post-shielding maximum dose rate of 60 mR/hr. The inspector could not determine from the contact dose rate measurements the extent of the actual dose rate reduction to workers in the RHR Suction line area.

- Pressurizer Spray Line. This shielding package consisted of an engineering load analysis for one layer of lead blankets, which translates to an additional 15 lb/ft load. This load analysis was evaluated for static, dynamic, thermal expansion, and seismic inertia loading and was determined acceptable. This was a good example of an engineering load analysis provided for temporary shielding, however it was performed for a singular load (as requested), rather than determining load limitations for various pipe sections and under various system operating modes. As before, the shielding effectiveness was measured with regard to contact pipe dose rates and did not provide the general field dose rates, which would allow dose values to be ascribed to each shielding effort after determination of area occupancy times. This particular shielding effort was recorded as 600 mR/hr contact with the pressurizer spray line before shield installation, down to 70 mR/hr at contact after shielding had been installed.

- Regenerative Heat Exchanger. A layer of 60 lead blankets were hung from a stand acting as shadow shielding for this source. No shield evaluation data were available for review.

- "B" Loop Crossover Piping. A layer of 10 lead blankets was laid on the crossover piping. The shielding design package specified the purpose of the shielding was to reduce dose rates to the "B" RCP ladder area. The post-shielding ALARA evaluation quoted crossover piping dose rates of 900 mR/hr at contact with the piping prior to shielding, down to 170 mR/hr at contact after shield installation. This shield package included the complete radiation survey data from which the inspector could determine the shield effectiveness relative to the stated shield design purpose, namely to shield the "B" RCP ladder area. Prior to shielding, the ladder area measured 80 mR/hr and after shielding 70 mR/hr was the result. Most of the other shield packages did not define the exposure reduction target.

- Chemical and Volume Control System (CVCS) Valve 427. This high radiation source was located on the "B" RCP platform and the shield was designed to reduce dose rates to the "B" RCP and "B" steam generator platforms. Eight lead blankets were used with measured contact dose rate measurements of 750 mR/hr before shielding and 40 mR/hr after shielding placement. The inspector reviewed the available radiation survey data and determined the pre-shielding dose rate values to range between 24 and 40 mR/hr, which were reduced to between 16 and 30 mR/hr after shielding installation.

- Pressurizer Spray Valves. A total of 15 lead blankets were used to shield these valves. A contact dose rate reduction of 560 mR/hr down to 170 mR/hr was recorded. The inspector reviewed surveys and determined a 140 mR/hr general field reading was reduced to 70 mR/hr.

The inspector reviewed 13 other shielding packages, more examples of those described above. In general, the shielding packages were preplanned and coordinated with the technical engineering group for approval, which was an improvement from the previous outage. General field dose rates were typically reduced by 50%. The engineering approvals for the outage shielding designs usually indicated that the shielding resulted in an insignificant additional load on piping systems. There was one example of a load analysis performed for the outage. In most cases, additional pipe loading was not a limitation. The shielding packages reviewed consisted of an initial, one-time request for one layer of lead blankets. Due to the complex nature of radiation source contributions inside containment, final shielding design determination is generally an iterative process based on empirically derived survey data. The need for additional shielding for any one particular shielding design was apparently not evaluated, as there were no examples of repetitive load analyses requested from engineering.

In addition, there was no evaluation of the outage workforce time to be expended in various areas of the plant to determine where the maximum dose benefit might be derived from shielding efforts, and no determination of shielding priorities. The development of a cost-versus-benefit methodology would provide a basis for determining the appropriate amount of effort to be assigned in each case to ensure exposures were as low as is reasonably achievable. During this outage, there was not enough shielding installation follow-up to allow a determination of shielding effectiveness, which also requires data input of actual outage workforce time spent in the shielded areas.

It should be noted that the pressurizer spray line shielding package was designed for permanent shielding bracket installation, which will allow reuse during future outages. The licensee has an approved lead blanket storage location inside containment that allows for more expedient shielding erections during outages. The licensee has implemented an expanded ALARA program over previous years. It is a noted strength that the Technical Services Group has dedicated onsite engineering support during outages to support minor modifications and shielding design approvals. Generally, the licensee has a good ALARA tracking and shielding program. Effective dose reductions have been made in this area. However, further enhancements and ALARA program developments should be considered.

8.3 Source Term Reduction

The inspector questioned the licensee about source term reduction activities at Ginna Station. Apparently since 1983, when steam generator channel heads were chemically decontaminated, only minor source term reduction actions have been implemented. During this outage, there were no significant source term reduction activities. The licensee is studying the feasibility of conducting a full reactor coolant system chemical decontamination in conjunction with steam generator replacement scheduled in 1996, and the feasibility of chemically decontaminating the regenerative heat exchanger.

9.0 Exit Meeting

The inspector met with licensee representatives at the end of the inspection, on April 16, 1993. The inspector reviewed the purpose and scope of the inspection and discussed the findings.